NON-PUBLIC?: N
ACCESSION #: 9508290188
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Haddam Neck PAGE: 1 OF 6

DOCKET NUMBER: 05000213

TITLE: Manual Reactor Trip and Safety Injection Due to Loss of
Main Feed Pump
EVENT DATE: 07/27/95 LER #: 95-016-00 REPORT DATE: 08/22/95

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(i), 50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: J. DeLawrence, Supervisor TELEPHONE: (203) 267-2556
Engineering Programs

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: SJ COMPONENT: M MANUFACTURER: W120
X AB LIC F180
B BA FI R290
B EL 52 W120
REPORTABLE NPRDS: Y
Y
N
Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On July 27, 1995, at approximately 0155 hours, with the plant in Mode 1 at 100 percent power, control room operators received a 4160 volt ground alarm. At 0156 the 'B' main feed pump motor breaker tripped resulting in steam flow/feed flow mismatch alarms on all four steam generators. At 0157 the operators manually tripped the reactor and turbine. Automatic initiation of the auxiliary feedwater system and the subsequent cooldown caused pressurizer level to decrease to less than 13 percent and safety injection was manually initiated. After restoring pressurizer level, safety injection was terminated at 0230 hours and the plant was stabilized in Mode 3 (hot standby). The cause of the event was degraded insulation in the main feed pump motor stator believed to be due to heat and the age and design of the motor. Corrective action consisted of installing a spare stator. In addition the "A" main feed pump was tested and found satisfactory. At 0510, with the plant in Mode 3, operators failed to obtain a loop boron sample before restarting the No. 1 reactor coolant pump and opening the affected loop cold leg isolation valve which constituted a violation of the Technical Specifications. The cause was operator error. Corrective action consisted of discussing with the crew the need to follow the Technical Specification requirements for sampling.

END OF ABSTRACT

BACKGROUND INFORMATION

The main feedwater system (EIIS Code:SJ) contains two electric main feedwater pumps (EIIS Code: P) each rated at 9600 gpm at a rated pressure of 1500 psig and associated motor (EIIS Code:M) rated at 4500 HP - 546 amp - 4000 volts. All four feedwater regulating valves (EIIS Code: FCV) go full open on a reactor trip and closed on a reactor coolant system (RCS)(EIIS Code:AB) Tavg less than 545 degrees F. The feedwater regulating valves will also go closed on a safety injection actuation signal (EIIS Code:JE). The auxiliary feedwater system (EIIS Code:BA) is used to remove residual heat from the reactor core when normal feedwater is unavailable. Each turbine driven auxiliary feedwater (AFW) pump is rated at 450 gpm, at 1050 psig. The AFW system automatically actuates due to either 1) both main feed pump circuit breakers (EIIS Code: 52) opening or 2) two of four steam generator wide range levels less than 48 percent on train A or B with a 30 second time delay. The auto actuation feature (EIIS Code:JE) has to be reset at the main control board to allow operators to manually control auxiliary feedwater flow.

Technical Specification 4.4.1.11.2 (Idled Loop Startup) states:

"Within 30 minutes prior to opening the idled loop cold leg stop valve, the idled loop shall be determined to have a boron concentration greater than or equal to the boron concentration required to meet the SHUTDOWN MARGIN requirements...If an idled loop is being started within 30 minutes after a reactor trip, this surveillance requirement may be waived if the cold leg loop stop valve is closed for less than 15 minutes".

EVENT DESCRIPTION

On July 27, 1995, at approximately 0155 hours, with the plant in Mode 1 at 100 percent power, control room operators received a 4160 volt ground alarm. An operator was sent to the 'A' switchgear room to investigate. At 0156 the 'B' main feed pump motor breaker tripped resulting in steam flow/feed flow mismatch alarms on all four steam generators. At 0157 the operators manually tripped the reactor and turbine and entered emergency response procedure E-O, "Reactor Trip or Safety Injection. Automatic initiation of the auxiliary feedwater system and the subsequent RCS cooldown caused pressurizer level to decrease to less than 13 percent . Safety injection was manually initiated as required by emergency response procedure E-O. After restoring pressurizer level, operators entered

emergency response procedure ES-1.1, "SI Termination". Safety injection was terminated at 0230 hours and the plant was stabilized in mode 3 (hot standby). During the transient pressurizer level reached a minimum of 12.1% and a maximum of 97.5%. Pressurizer pressure reached a minimum of 1742 psig and a maximum of 2119 psig. At 0510, with the plant in Mode 3, operators failed to obtain a reactor coolant loop boron sample before restarting the No. 1 reactor coolant pump (RCP) and opening the affected loop cold leg isolation valve. This resulted in violating Technical Specification 3.4.1.1.11.2, "Idled Loop Startup". Since the loop was not returned to service within 30 minutes after the reactor trip the boron sample was required to be taken.

The following equipment did not respond as expected during this event:

1. The pressurizer level controller was in remote-auto prior to the transient but appears to have switched to remote-manual during the transient. Pressurizer level exceeded the program value of 25% however the charging flow did not change. The operator took manual control to minimize the charging rate.

2. With both the AFW pumps and a main feedwater pump running following the trip control room operators noted erratic auxiliary feedwater flow indication with the flow indicators going to zero.

3. Following receipt of the safety injection actuation signal the red open indication light for SI-MOV-861D (high pressure safety injection to loop 4) did not come on. However, based on the green closed light going out, it was concluded that the valve went to its open position.

4. During the recovery from the reactor trip one of two generator output breakers (14B-3T-2) failed to reclose on the 345 KV ring bus.

CAUSE OF THE EVENT

Investigation of the 4160 volt ground determined that the ground was caused by a motor phase to ground on the 'B' main feed pump motor followed by a phase to phase ground causing the feed pump breaker to open due to instantaneous overcurrent. The cause of the event was degraded insulation in the main feed pump motor stator believed to be due to heat and the age and design of the motor.

The cause of the pressurizer level controller anomaly is unknown. The controller was put through a series of tests but the response noted during the trip could not be duplicated.

The cause of the erratic auxiliary feedwater flow indication was an intrinsic design characteristic of the vortex shedding flow meter. If the actual flow exceeds the maximum range of the flow meter (600 gpm) the meter will re-zero causing the main control board indication to read zero.

The cause of the red indicating light for SI-MOV-861D not coming on is unknown. A suspected cause is improper installation of the indicating bulb, or poor contact between the socket and bulb prior to the event.

The cause of the failure of 14B-3T-2 to close was breaker vibration during its closing cycle causing the relay contact to reopen the breaker.

The cause of the failure to take a boron sample prior to returning reactor coolant loop I to service was operator error in the interpretation of the requirements contained in normal operating procedure NOP 2.4-2, "Reactor Coolant Pump Operation".

SAFETY ASSESSMENT

This event is reportable under 10CFR50.73 (a) (2) (iv) as any event or condition that results in manual or automatic actuation of any Engineered Safety Feature including the Reactor Protection System. This event resulted in the manual actuation of a reactor trip, automatic actuation of the auxiliary feedwater system and manual actuation of the safety injection actuation system. This event is also reportable under 10CFR50. 73 (a) (2) (i) (B) as a condition prohibited under the plant's Technical Specifications due to the failure to take a reactor coolant loop boron sample prior to returning loop 1 to service.

The loss of normal feedwater flow is an analyzed event for the Haddam Neck plant. The analysis assumes a total loss of normal feed flow in addition to the failure of one auxiliary feedwater pump as the single active failure The analysis also assumes lifting steam generator safety valves. No credit is taken for the steam dump to the condenser.

The event resulted in the automatic actuation of both auxiliary feedwater pumps due to low level in two of four steam generators. Steam dump to the condensers operated properly and no steam generator safety valves lifted. Failure of the pressurizer level controller had the potential of filling the pressurizer and challenging the lift setpoints of the pressurizer safety valves. The auxiliary feedwater flow indicator anomaly and the light indication problem with SI-MOV-861D had minimal safety significance on the event. Other indication was available to show that feed flow was reaching the steam generators and that SI-MOV-861D went to its open position.

Based on equipment availability and operator response to this event, the safety significance was less severe than already assumed in the safety analysis for the loss of normal feedwater flow transient.

The failure to take the boron sample prior to returning an idled loop to service could potentially result in a boron dilution event. However, since the loop was not isolated from the reactor coolant system and 2 of the four reactor coolant pumps were running, backflow through the idled loop still existed. With backflow established, the boron concentration within loop 1 was maintained at the same level as the rest of the reactor coolant system. Shutdown margin was not challenged as a result of not taking the boron sample because of both backflow through the loop and the negative reactivity present due to post trip xenon concentrations. Based on these conditions the safety significance was minimal.

CORRECTIVE ACTION

Corrective action consisted of installing a spare stator in the 'B' main feed pump. In addition the 'A' main feed pump was tested and found satisfactory. Long term corrective action is to evaluate options for improving the reliability of the main feed pump motors such as replacing the motors with a new design.

The program card in the pressurizer level controller was removed, a new card was installed and the controller was calibrated. The card has been sent to the vendor for further investigation and failure analysis. In addition all control room operators were informed of the potential for this anomaly to occur.

Long term corrective action for the auxiliary feedwater flow indicators is to evaluate the feasibility of modifying the system to prevent erratic indication. In addition all control room operators were informed of the potential for this anomaly to occur.

The indicating light for SI-MOV-861D was removed and tested satisfactory. The valve was tested in accordance with its surveillance procedure and the valve operated satisfactorily including the indicating lights.

The relay for 14B-3T-2 was relocated to prevent it from being affected by the vibration of the breaker closing.

Corrective action for the failure to take a boron sample consisted of discussing with the crew the need to follow the Technical Specification requirements for sampling.

ADDITIONAL INFORMATION

Component Manufacturer Model No

'B' Feed Pump Motor Westinghouse SIN 1S-719154
Type - CS

Pressurizer Level Controller Foxboro N-2CDA-N1+HS3

AFW Flow Indicator Robert Shaw Control 237-C1

Circuit Breaker (4160 v.) Westinghouse 50 DHP 250

PREVIOUS SIMILAR EVENTS

None.

ATTACHMENT TO 9508290188 PAGE 1 OF 1

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
362 INJUN HOLLOW ROAD o EAST HAMPTON, CT 06424-3099

August 22, 1995
Re: 10CFR50.73(a)(2)(i)

10CFR50.73(a)(2)(iv)

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Reference: Facility operating License No. DPR-61
Docket No. 50-213
Reportable Occurrence LER 50-213/95-016-00

This letter forwards the Licensee Event Report 95-016-00, required to be submitted, pursuant to the requirements of the Haddam Neck Plant's Technical Specifications.

Very truly yours,

F. R. Dacimo
Vice President

FRD/eda

Attachment: LER 50-213/95-016-00

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. William J. Raymond
Sr. Resident Inspector
Haddam Neck

1028-3 REV. 2-91

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